ACCESSION #: 9906180061

NON-PUBLIC?: N

LICENSEE EVENT REPORT (LER)

FACILITY NAME: BYRON STATION, UNIT 1 PAGE: 1 OF 5

DOCKET NUMBER: 05000454

TITLE: AUTOMATIC REACTOR TRIP DUE TO HUMAN ERROR DURING

SURVEILLANCE PROCEDURE

EVENT DATE: 05/13/99 LER #: 99-003-00 REPORT DATE: 06/11/99

OTHER FACILITIES INVOLVED: DOCKET NO: 05000

OPERATING MODE: 1 POWER LEVEL: 100

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR SECTION:

50.73(a)(2)(iv)

LICENSEE CONTACT FOR THIS LER:

NAME: Brad Adams, TELEPHONE: (815) 234-5441

Regulatory Assurance Manager x2280

COMPONENT FAILURE DESCRIPTION:

CAUSE: x SYSTEM: SB COMPONENT: VOP MANUFACTURER: A499

D JB RV D245

REPORTABLE NPRDS: Y

Y

SUPPLEMENTAL REPORT EXPECTED: NO

ABSTRACT:

On May 13, 1999, at 0810 hours, with Byron Station Unit 1 at 100% power, an

Instrument Maintenance Technician (IM) inadvertently caused a Reactor Trip System (RTS) automatic actuation. This occurred during the performance of the Technical Specification Surveillance Requirement (SR) for the Power Range Neutron Flux Instrumentation channel. The SR being performed was the 92 day frequency Channel Operational Test (COT). The IM mistakenly removed instrument fuses from a channel adjacent to the channel he had placed in test for the surveillance procedure. This action caused the two of four coincidence logic to be satisfied for the generation of an automatic reactor trip. Operators responded using appropriate procedures. The cause of the trip was a human performance error by the IM conducting the surveillance procedure. There were no safety consequences impacting plant or public safety as a result of the event. The FITS functioned as designed. Corrective actions include appropriate Management action with the IM involved and re-enforcement of human performance expectations with all Maintenance Department personnel. There was one previous occurrence involving an RTS actuation due to a human performance cause reported in Byron Station Licensee Event Report 454-96-17. This event is reportable in accordance with 10 CFR 50.73 (a)(2)(iv).

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A. PLANT CONDITIONS PRIOR TO EVENT:

Byron Station Unit 1 is a four-loop pressurized water reactor nuclear steam supply system with a turbine generator supplied by Westinghouse Electric Corporation. At the time of the event, the Unit 1 Boron Dilution Protection System (BDPS) was inoperable as described in Byron Station Licensee Event Report 454-98-20. BDPS is required by Technical Specifications to be operable in Mode 3 (i.e., Hot Standby) through Mode 5 (i.e., Cold Shutdown). No other structures, systems or components (SSCs) were inoperable at the start of the event that contributed to the event

Event Date/Time: May 13, 1999 / 0810

Unit 1 Mode 1 - Power Operations

Reactor Power - 100%

Reactor Coolant System (RCS) [AB] Status:

Temperature - Normal Operating Temperature

Pressure - Normal Operating Pressure

B. DESCRIPTION OF EVENT:

On May 13, 1999, at 0810 hours, a non-licensed Instrument Maintenance Technician (IM) inadvertently caused a Reactor Trip System (RTS) automatic actuation during the performance of the Technical Specification Surveillance Requirement (SR) for the Power Range Neutron Flux Instrumentation (NR) [IG] channel. The SR was the 92 day frequency Channel Operational Test (COT).

Byron Station's Technical Specifications require four NR channels to be operable as part of the limiting condition for operation for the RTS instrumentation. These instrument channels provide a reactor trip function for both high and low neutron flux conditions. These functions require a two of four channel logic coincidence to actuate the FITS. To perform the COT of the NR instrumentation, one NR channel at a time is placed in the test condition that, in effect, causes a one of four coincidence logic input to the RTS for high neutron flux during the performance of the surveillance procedure. Therefore, an actual or spurious condition causing the high flux setpoint to be exceeded on any of the remaining three operable NR channels will satisfy the two of four coincidence logic and cause an RTS actuation. It is normal practice to perform the COT on all four of the Power Range monitors (i.e., NR 41, 42, 43 and 44) one at a time in sequence.

The NR instrumentation tested during the performance of the NR COT is located in Main Control Room cabinets, which are physically connected in series, NR 41 through NR 44. The COT is performed using station procedure Byron Instrument Surveillance Requirement (BISR) 3.1.6-200, "92 Day Surveillance Calibration of Power Range Nuclear Instrumentation System." The procedure requires IM actions to be performed both in front and rear of these cabinets.

On May 13, 1999, at 0018 hours, the Technical Specification Action for channel NR 41 was entered and an IM began the first of the four COT activities. He had successfully completed calibration of the second channel, NR 42, by the end of his shift. Later the same morning, on May 13, 1999, the night shift IM turned over the activity of completing the COT for the remaining two NR channels to the day shift IM.

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B. DESCRIPTION OF EVENT (cont):

At 0736 hours, the Technical Specification Action for NR 43 was entered and the day shift IM began the COT for NR 43. During the surveillance procedure, the IM realized he did not have the correct test equipment connector to execute one of the procedure steps in the

COT and sent his IM assistant to obtain it. This was just after performing a procedure step in the rear of the NR 43 cabinet. During this pause in the procedure he questioned himself as to whether he had properly executed a previous step of removing the instrument power fuses, which are located on the front of the cabinet. He was now standing in front of the NR 42 cabinet. He mistakenly looked at the NR 42 instrument fuses, which were installed, and erroneously confirmed his suspicion that he missed the previous step. He assumed the NR 42 fuses were the NR 43 fuses and, without further independent corroboration, removed them. The fuses for NR 43 had been, in fact, properly removed in accordance with the surveillance procedure.

By removing the instrument fuses from NR 42, with channel NR 43 in a test condition, the two of four logic coincidence was satisfied for the NR high neutron flux. The RTS actuated as designed and the reactor was automatically shutdown. A reactor trip also causes an automatic turbine trip [TA] and Main Feedwater [JB] isolation. These functions occurred as designed. Licensed Operators responded to the reactor trip using appropriate procedures. A reactor trip from power also causes an immediate entry into Mode 3 and as a result, Operators entered the Technical Specification Action for BDPS inoperability. This required isolation of boron dilution flow paths within one hour. This was accomplished at 0901 hours, on May 13, 1999.

As an expected response to reactor trip from 100% power, the Steam

Generator water levels dropped below the low level setpoint for an automatic Auxiliary Feedwater [BA] actuation. This resulted in the automatic initiation of both trains of the Auxiliary Feedwater system. This is considered an Engineered Safety Feature (ESF) actuation. With the turbine isolated, RCS temperature is maintained via 12 Main Steam (MS) [SB] Dump valves (i.e., 1MS004A through M) which relieve to the Condenser. These MS dumps valves opened and modulated as designed to relieve steam and maintain RCS temperature. However, the 1MS004K steam dump valve failed to close after properly opening. This caused an RCS cooldown to below the automatic block of MS Dumps valves setpoint of 550 degrees Fahrenheit (F). Operators quickly recognized this condition and took action to close the manual block valve to isolate the 1MS004K valve. The minimum RCS temperature reached was approximately 547 degrees F. After isolation of the MS dump valve, 1MS004K, RCS temperature returned to the expected no-load FICS average temperature of 557 degrees F.

The radiation monitor for the Unit 1 Component Cooling [CC] System momentarily spiked high coincident with the reactor trip, causing an automatic surge tank vent path isolation. This does not represent an ESF actuation. This is because the signal causing the isolation is not processed through the ESF protection circuitry. This vent path was re-opened at 1002 hours after confirmation that the radiation level was normal and the isolation was spurious.

In addition, the secondary plant pressure transient resulting from the automatic feedwater isolation caused numerous feedwater heater relief valves to lift. Eleven of these valves failed to re-seat fully.

These valves relieve to the turbine building drain system. The failure of these valves resulted in Operators having to isolate the feedwater heater strings.

At 0847 hours on May 13, 1999, a NRC Emergency Notification System telephone call was initiated in accordance with 10 CFR 50.72 (b)(2)(ii). This event is also reportable in accordance with 10 CFR 50.73 (a)(2)(iv). A prompt investigation into the root cause of the reactor trip was initiated.

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C. CAUSE OF EVENT:

The root cause of the reactor trip was determined to be a human performance error by the IM performing the COT of channel NR 43.

The cause of the failure of the 1MS004K valve was a broken control feedback arm on the valve's positioner. This is believed to have occurred during the valve's response during the reactor trip.

The cause of the CC System surge tank isolation was attributed to a momentary power transient on the electrical bus supplying the radiation monitor. A reactor trip causes an automatic bus transfer from the radiation monitor's normal non-safety power supply to the off-site power source. This momentary transient causes the radiation monitors to fail in the safe mode, i.e., an assumed high radiation level signal. This conservatively resulted in the isolation of the vent path.

The feedwater heater relief valves are thermal relief valves designed to relieve slow pressure increases while the heater is isolated. They are not designed to respond to sudden large pressure increases. As a result of its design, the valves lifted and reseated numerous times until it eventually failed.

D. SAFETY ANALYSIS:

The RTS is designed to automatically keep the reactor from operating in an unsafe condition by shutting down the reactor whenever prescribed limits are exceeded. This actuation of the RTS was a result of human error during the performance of a surveillance and not due to any actual reactor safety parameter being exceeded. There were no actual safety consequences impacting plant or public safety as a result of the event. The reactor trip system functioned as designed.

The Main Steam Dump System is non-safety related and is the preferred method of RCS temperature control in Mode 3 because it preserves condensate inventory. If this system had failed to function as designed, there are safety-related systems that were fully capable of maintaining RCS temperature. The RCS cooldown below the no-load average temperature of 557 degrees F was quickly recognized by plant

Operators and responded to appropriately. The RCS average temperature was recovered before dropping below 547 degrees

The CC System surge tank allows for water expansion due to temperature increases. A radiation monitor continuously monitors the CC System for in-leakage from the RCS or Residual Heat Removal [BP] system. A high radiation level signal initiates an automatic isolation of the surge tank vent line to prevent the release of airborne radioactivity. The isolation of the vent path for the CC System Surge tank had no safety consequences and was a conservative action. The actuation signal was spurious due to a momentary power transient and not in response to an actual radiation leak. This situation did not aggravate nor hamper the plant Operators in responding to the reactor trip.

Failure of the feedwater heaters relief valves to close as designed had no adverse nuclear safety consequences. These valves are non-safety related and have no impact on safety related SSCs needed to recover from reactor trip.

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E. CORRECTIVE ACTIONS:

Appropriate Management action was taken to address inappropriate personal performance during the performance of the Surveillance. The action taken was commensurate with the seriousness of the issue.

A Byron Station Maintenance Department "stand down" of work activities was completed following the event to discuss the event and to reinforce to all personnel the seriousness of this event and the need to always use error reduction techniques.

Maintenance supervision will continue to re-emphasize to their technicians the expectation that all personnel use error reduction techniques.

Surveillance procedure BISR 3.1.6-200 and related BISR 3.1.6-202, "Surveillance Calibration of Nuclear Instrumentation System Power Range N41 - N44," have been revised to restrict the removal of NR fuses only by licensed plant operators.

The feedback arm on the 1MS004K steam dump valve was repaired.

The eleven feedwater heater relief valves were all replaced prior to unit startup.

F. RECURRING EVENTS SEARCH AND ANALYSIS:

Byron Station LER 454-96-17, "Unit 1 Reactor Trip Due to Personnel Error During Surveillance Activities." This report describes an event involving a plant operator, who during the performance of a surveillance procedure, placed a lever-actuating tool on the Turbine Protection System "Trip" lever instead of the intended "Test" lever. While trying to rectify this error, he inadvertently caused a turbine trip/reactor trip.

G. COMPONENT FAILURE DATA:

1) Bailey Positioner, Model AV-1

2) Consolidated Valve, Model 1990C1205

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Commonwealth Edison Company

Byron Generating Station

4450 North German Church Road

Byron, IL 61010-9794

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June 11, 1999

LTR: BYRON 99-0064

FILE: 3.03.0800

U.S. Nuclear Regulatory Commission

ATTN: Document Control Desk

Washington, D.C. 2055S-0001

Byron Station Unit 1

Facility operating License No. NPF-37

NRC Docket No. STN 50-454

SUBJECT: Licensee Event Report (LER)99-003

Dear Sir:

Enclosed is the LER for the Unit 1 Reactor Trip Event on May 13, 1999.

This event is reportable in accordance with 10 CFR 50.73(a)(2)(iv). If you need any additional information concerning this report, please contact Mr. Brad Adams, at (815)234-5441, extension 2280.

Sincerely,

/s/

William Levis

Site Vice President

Byron Station

WL/JL/dk

Enclosure: Licensee Event Report No. 99-003

cc: James E. Dyer, NRC Region III Administrator

NRC Senior Resident Inspector - Byron Station

INPO Record Center

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